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SUBJECT: Reactor Physics Calculations for the MSRE

TO: Distribution

FROM: C. W. Nestor, Jr.

Summary

This memorandum is a compilation of results obtained to date from a number of reactor-physics calculations for the molten salt reactor experiment (MSRE). Included are one-dimensional multigroup and two-dimensional two-group calculations of critical mass, flux and power density distributions; gamma heating in the core can, reactor vessel and core support grid; drain tank criticality; and an estimate of the beta, gamma and delayed neutron dose rates due to fission products in the fuel contained in the pump bowl.

For a cylindrical core 54 inches in diameter and 66 inches high, graphite-moderated with 8 volume percent fuel salt, the calculated critical uranium loading is 0.76 mole percent uranium (93.3% U-235), which is equivalent to a critical mass of 16 kilograms. At a reactor power of 10 megawatts, the peak power density in the core assuming a homogeneous mixture of fuel salt and graphite is 10 watts/cm^3, the average power density is 4 watts/cm^3. The computed peak thermal flux is 7.3×10^{13} n/cm^2 sec and the average is 2.5×10^{13} n/cm^2 sec. Gamma heating produces a power density of 0.2 watts/cm^3 in the core wall at the midplane and 0.4 watts/cm^3 in the support grid at the bottom of the core at the reactor center line.

DISCLAIMER

Core Calculations

The model of the reactor considered is shown in Fig. 1. The fuel salt is a mixture of 64 mole percent LiF (0.03% Li^6), 31 mole percent BeF_2, 4 mole percent ThF_4, and about 1 mole percent UF_4 (93.3% U^{235}). The core consists of 2-inch square graphite blocks with 0.1 inch x 1.5 inch fuel channels between adjacent blocks (Figure 2); in the calculations the core was assumed to be a right circular cylinder 54 inches in diameter and 66 inches high containing a homogeneous mixture of 8 volume percent fuel salt and 92 volume percent graphite. The regions designated "Grid 1" and "Grid 2" in Figure 1 are homogeneous mixtures of fuel salt, graphite and INOR-8; these regions are representations of the support grid and the associated fuel salt and graphite. The core can and reactor vessel are INOR-8 and there is a 1-inch annulus filled with fuel salt between the can and the vessel. The region designated "Header" represents the fuel salt in the bottom of the vessel. Results of one-dimensional multigroup calculations[1] along the center line and midplane were used to generate two-group constants for the two-group, two-dimensional calculations;[2] power density distributions calculated with the two-group method are shown in Figures 3 and 4 for a reactor power of 10 megawatts. Results for the 54 x 66, 8-volume-percent-fuel salt reactor are tabulated in Table 1; results for a number of reactors of different diameters are shown in Figure 5, and neutron balances are listed in Table 2.

Table 1. Results of Nuclear Calculations for MSRE

Core size: 54 inches diameter, 66 inches height right circular cylinder

Power: 10 megawatts

Peak core power density: 10.2 watts/cm^3

Mean core power density: 4.04 watts/cm^3

Fuel salt volume fraction 8%

Critical mass in core: 15.7 kg.

Critical fuel concentration: 0.76 mole percent U (93.3% U^{235})

Peak thermal flux: 7.3×10^{13} n/cm^2 sec

Mean thermal flux: 2.5×10^{13} n/cm^2 sec

Table 2. Neutron Balances for Reactors Having Various Diameters and Heights (basis: 100 neutrons absorbed in U^{235})

Absorptions:	$3\frac{1}{2}$' D x $5\frac{1}{2}$'H	4' D x $5\frac{1}{2}$'H	$4\frac{1}{2}$'D x $5\frac{1}{2}$'H	5'D x $5\frac{1}{2}$'H
Be	0.02	0.04	0.06	0.08
C	0.67	1.73	2.68	3.53
F	0.34	0.46	0.56	0.64
Li^6	0.91	2.33	3.61	4.74
Li^7	0.11	0.27	0.42	0.55
Th	12.97	17.90	21.25	24.01
U^{238}	1.86	1.05	0.78	0.65
Leakage:				
Fast	66.03	60.52	53.45	47.40
Thermal	4.54	11.65	15.94	18.43
$\overline{\eta}$	1.874	1.960	1.988	2.000

Table 2 - cont'd

(basis: 100 neutrons absorbed in U^{235})

Absorptions:	$3\frac{1}{2}$'D x 10'H	4'D x 10'H	$4\frac{1}{2}$'D x 10'H	5'D x 10'H
Be	0.03	0.06	0.09	0.11
C	1.22	2.46	3.82	4.72
F	0.40	0.54	0.68	0.77
Li^6	1.65	3.31	5.14	6.33
Li^7	0.19	0.39	0.60	0.74
Th	15.83	20.54	24.90	27.54
U^{238}	1.31	0.83	0.61	0.53
Leakage:				
Fast	64.00	54.94	45.57	40.11
Thermal	8.61	15.08	19.13	20.54
$\bar{\eta}$	1.933	1.981	2.005	2.014

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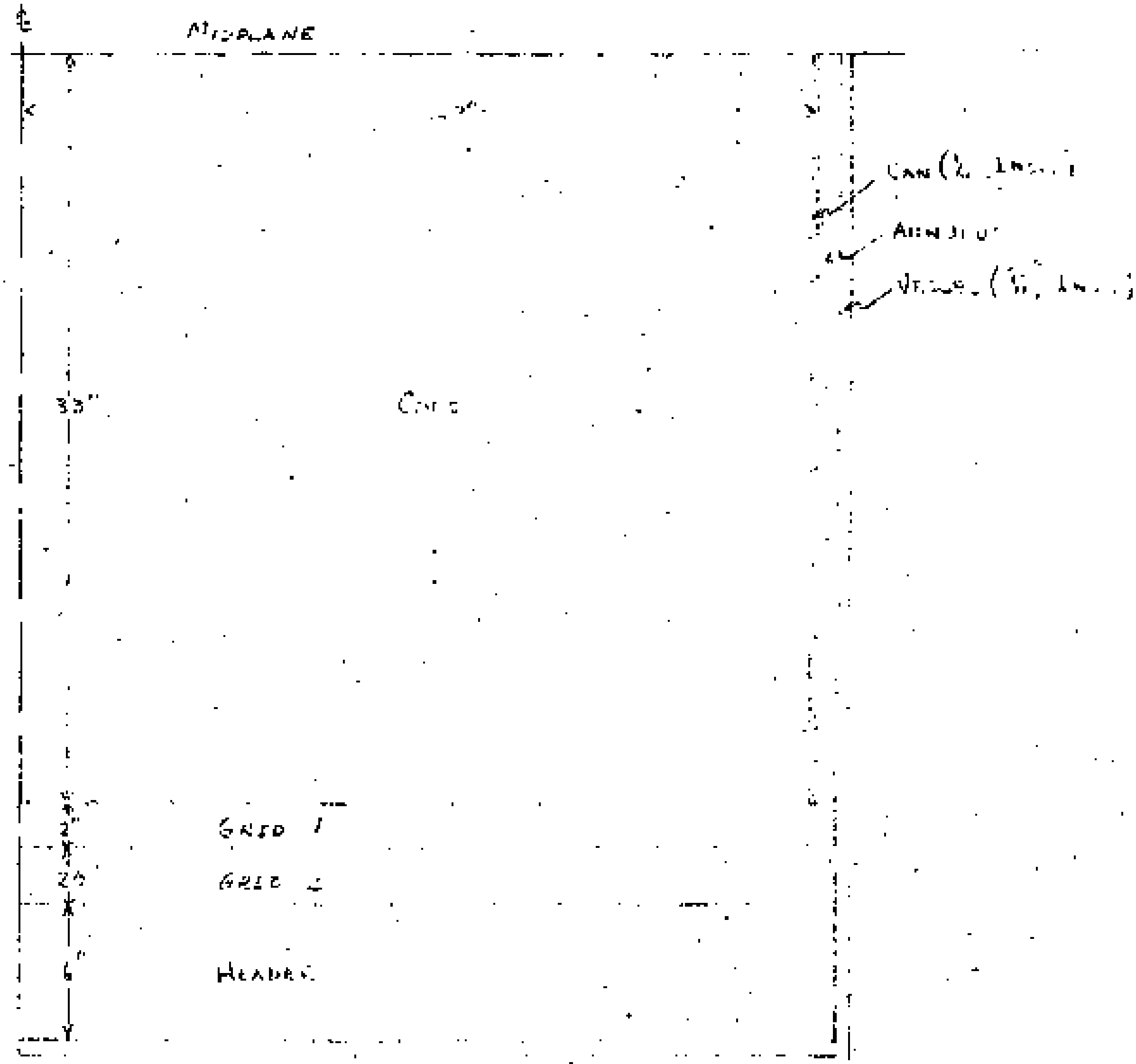

Fig 1. Reactor Model

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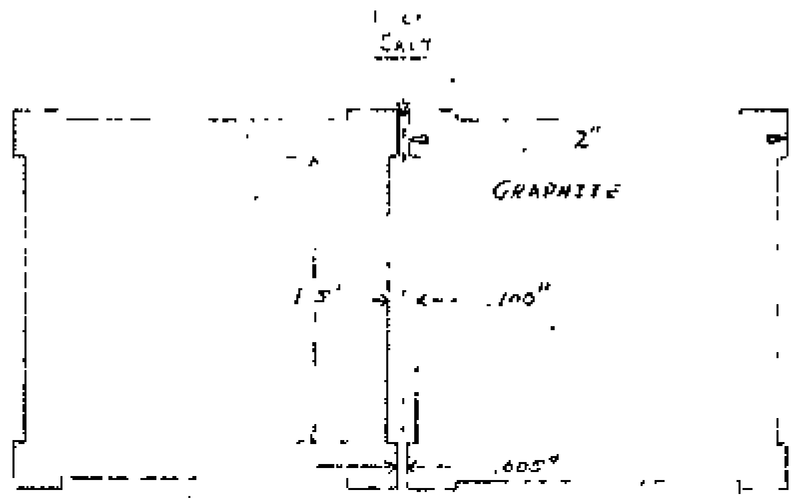

FIG 2. CORE CELL AND FUEL CHANNEL

RADIAL POWER DENSITY DISTRIBUTION

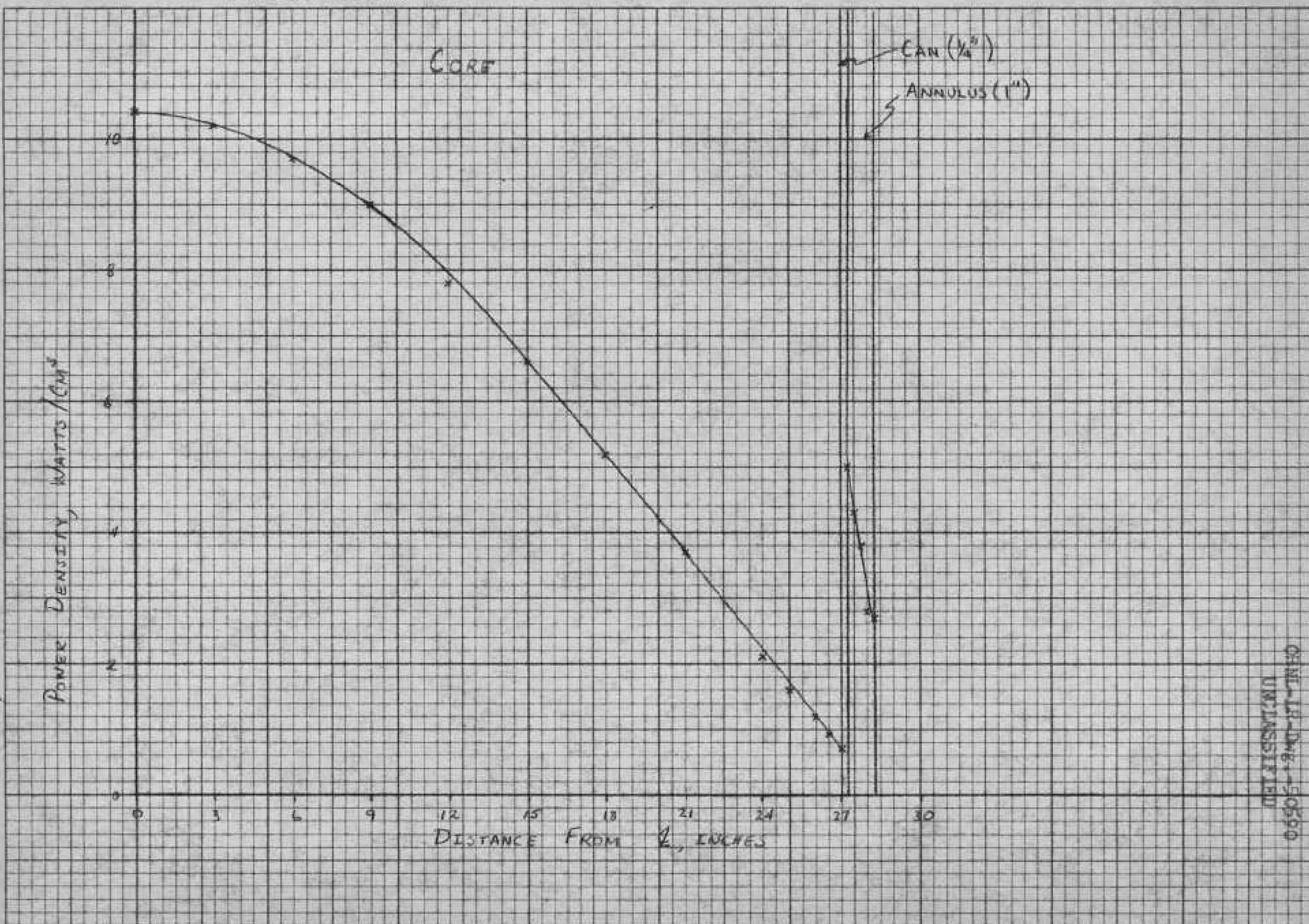

Fig. 3

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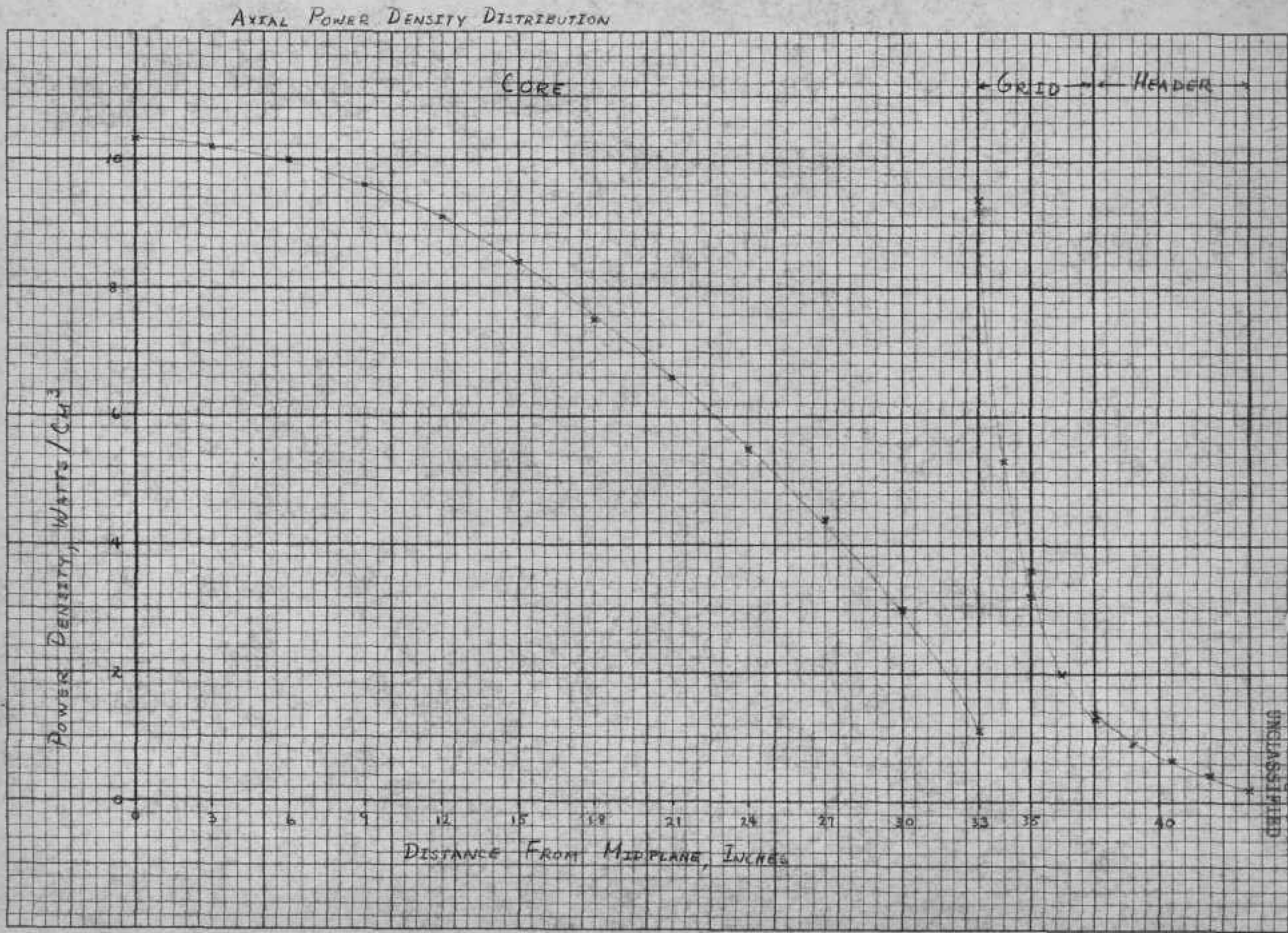

FIG. 4

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Fig. 8

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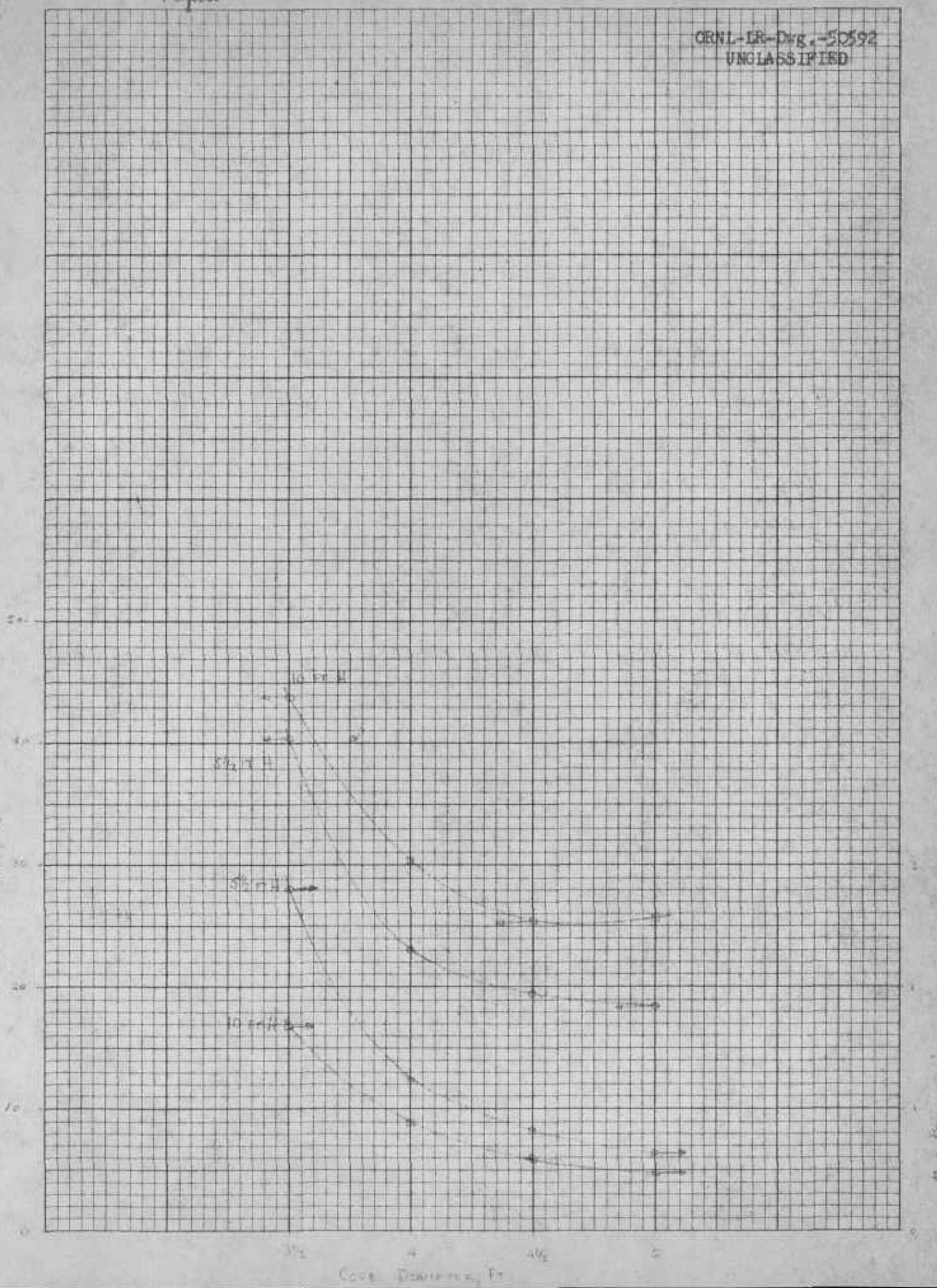
Figure 5. Critical Masses and Concentrations for Various Diameters



Drain Tank Criticality

The MSRE drain tank is a cylinder 5 feet in diameter and 5 feet high; when filled with fuel salt and surrounded by a 4-ft layer of water the tank has an effective multiplication constant (calculated with the multigroup 704 program GNU) of 0.77; without the layer of water, the effective multiplication constant is 0.44. Thus, even when reflected with an essentially infinite layer of water the tank is considerably subcritical.

Gamma Heating

Gamma heating calculations were done with the IBM 704 program Nightmare[3] using two-group two-dimensional neutron flux distributions calculated by the IBM-704 program Equipoise.[2] The results are tabulated in Table 3 for INOR-8 in the core can, pressure vessel and core support grid.

Table 3. Gamma Heating in the MSRE

Location	Heat generation, watts/gram	Heat generation, watts/cm^3 (ρ = 8.9 gm/cm^3)
Core can	0.024	0.21
Pressure vessel (inside)	0.022	0.20 } midplane
" " (outside)	0.015	0.13 } midplane
" " (center line)	0.042	0.38
Support grid (upper)	0.252	2.24
" " (lower)	0.197	1.75

Fission Product Activities in the Pump

Fission product activities of two general classes contribute to the radiation dose to pump lubrication. The first of these is the gaseous products and their descendants which collect in the gas volume; the second is the fission products (β, γ and delayed neutron emitters) which remain in the fuel liquid contained in the pump.

The computational model and a list of fission products for calculating the gas-borne activities was taken from the report concerning the ART by Stevenson.[4] The results for three different sweep rates and three different purge rates are shown in Figure 6, in terms of power released in the gas space, assuming that either the descendants are gaseous or that they are all plated out on the walls. Calculations for other combinations of sweep and purge rates and larger fission product chains may be done with an IBM-704 program prepared for this purpose; up to 100 fission products may be included.

The gamma source in the fuel salt contained in the pump may be estimated as follows. On the average, 7 decay gammas with a total energy of 5.5 Mev are emitted per fission. At equilibrium, the rate of decay must equal the rate of production; the number of gammas released per second must then be

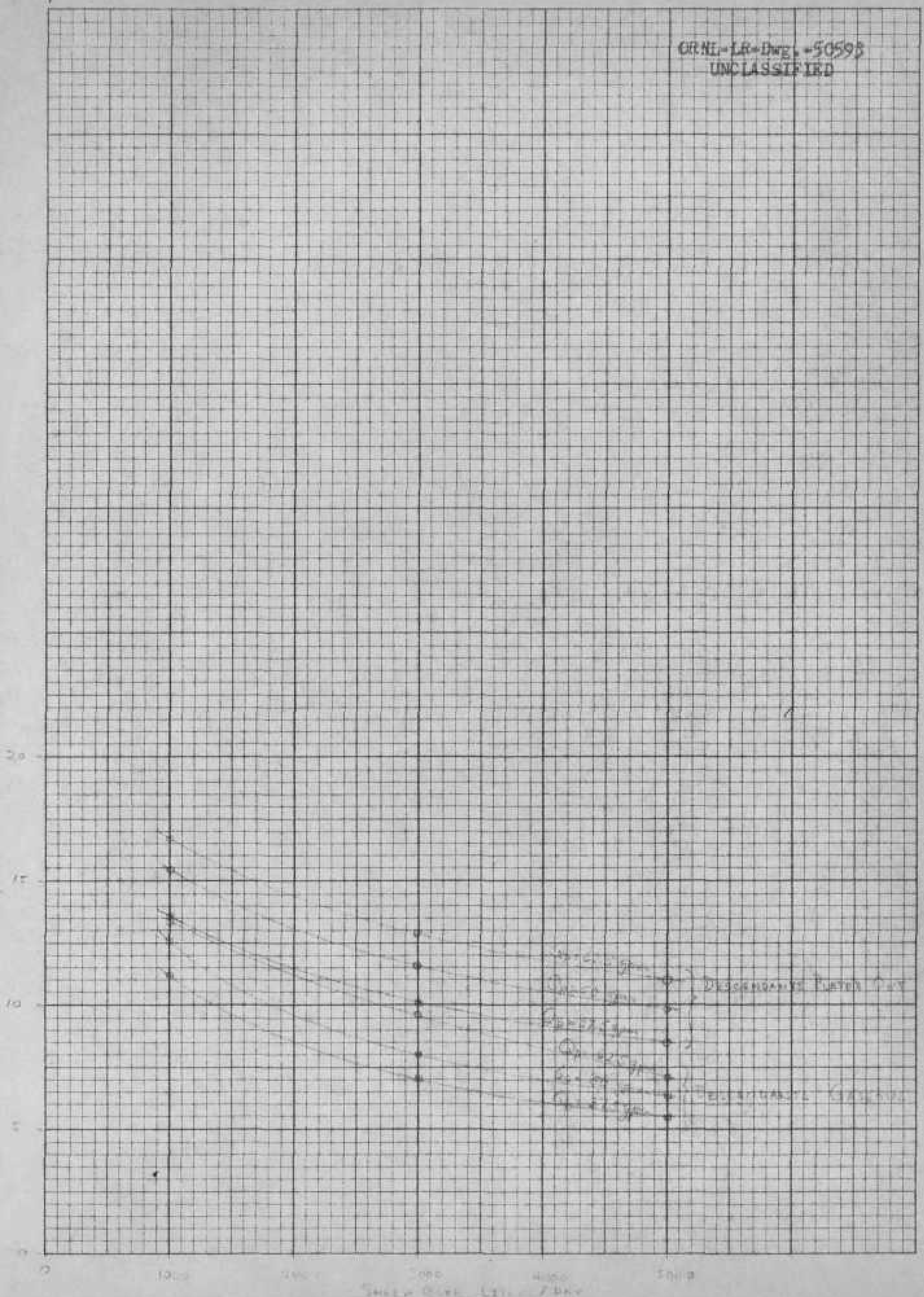
$$\frac{\text{gammas}}{\text{fission}} \times \frac{\text{fissions}}{\text{second}}$$

and the fraction of the total gamma activity which decays in the pump is

$$\frac{\text{volume of fuel in pump}}{\text{volume of fuel in system}} .$$

For a total power of 10 megawatts, the total decay gamma activity in the pump is given by

Figure 6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1000
2000
3000
4000
5000

$$I_\gamma = 10^7 \text{ watts} \times 3.38 \times 10^{10} \frac{\text{fissions}}{\text{watt sec}} \times 7 \frac{\text{photons}}{\text{fission}} \times \frac{3.5 \text{ ft}^3}{53 \text{ ft}^3}$$

$$= 1.6 \times 10^{17} \frac{\text{photons}}{\text{sec}}$$

and the volume source strength S_v is

$$S_v = 1.6 \times 10^{12} \frac{\text{photons}}{\text{cm}^3 \text{ sec}} .$$

Treating the photons as 1 Mev gammas and assuming the fuel pool in the pump is a disk 3 ft in diameter and 6 in. deep, the unshielded dose rate at 3 ft above the bottom of the pool is 1.7×10^7 Rad/hr, using the truncated cone approximation in TNX-7.[5]

The effective delayed neutron fraction $\overline{\beta}_i$ for each group of delayed neutrons is

$$\overline{\beta}_i = \beta_i \frac{1 - e^{-\lambda_i t_c}}{1 - e^{-\lambda_i t_L}}$$

where t_c is the core residence time, t_L the circulation time, λ_i the decay constant and β_i the yield of the ith delayed neutron group. At equilibrium the delayed neutron production rate must equal the production rate of delayed neutron precursors; we have for the reactivity per unit volume of the ith group of precursors (and thus for the source strength of the ith group of delayed neutrons)

$$\lambda_i C_i = \nu \overline{\beta_i} \Sigma_f \phi$$

where $\Sigma_f \phi$ is the fission rate per unit volume and ν the number of neutrons per fission.

Taking the total fuel volume as 53 ft^3, the core fuel volume as 10 ft^3, the system flow rate as 1250 gal/min and the delayed neutron data of Keepin et al[6] yields a source strength of

$$S_v \text{ (delayed neutrons)} = 1.52 \times 10^9 \text{ neutrons/cm}^3 \text{ sec.}$$

Assuming that the source is concentrated in a point 3 feet from an unshielded receptor the delayed neutron dose rate is $\sim 10^5$ rem/hr, less than 1% of the gamma dose rate from the decay gammas.

REFERENCES

1. C. L. Davis, J. M. Bookston, and B. E. Smith, GNU II - A Multigroup One-dimensional Diffusion Program for the IBM-704, General Motors Report GMR 101 (1957).

2. Melvin L. Tobias and T. B. Fowler, Equipoise - An IBM-704 Code for the Solution of Two-Group Neutron Diffusion Equations in Cylindrical Geometry, ORNL-2967 (1960).

3. M. P. Lietzke and M. L. Tobias, personal communication.

4. R. B. Stevenson, Radiation Source Strengths in the Expansion Chamber and Off-Gas System of the ART, CF 57-7-17 (1957).

5. C. W. J. Wende, The Computation of Radiation Hazards, Du Pont report TNX-7.

6. G. R. Keepin, T. F. Wimett, and R. K. Zeigler, "Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium and Thorium," J. Nucl. Energy 6, (1957).

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